June 19, 2001

Mr. J. B. Beasley, Jr.
Vice President
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Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Beasley:

Enclosed please find the Risk-Informed Inspection Notebook which incorporates the updated Significance Determination Process (SDP) Phase 2 Worksheets that inspectors will be using to characterize and risk-inform inspection findings. This document is one of the key implementation tools of the reactor safety SDP in the reactor oversight process and will also be publically available through the Nuclear Regulatory Commission (NRC) external website at http://www.nrc.gov/NRC/IM/index.html.

The 1999 Pilot Plant review effort clearly indicated that significant site-specific design and risk information was not captured in the Phase 2 worksheets forwarded to you last spring. Subsequently, a site visit was conducted by the NRC to verify and update plant equipment configuration data and to collect site-specific risk information from your staff. The enclosed document reflects the results of this visit.

The enclosed Phase 2 Worksheets have incorporated much of the information we obtained during our site visits. The staff encourages further licensee comments where it is identified that the Worksheets give inaccurate significance determinations. Any comments should be provided to the Document Control Desk, with a copy to the Chief, Probabilistic Safety Assessment Branch, Nuclear Reactor Regulation. We will continue to assess SDP accuracy and update the document based on continuing experience.

While the enclosed Phase 2 Worksheets have been verified by our staff to include the site-specific data we will continue to assess its accuracy throughout implementation and update the document based on comments by our inspectors and your staff.

If you have any questions, please contact me at 301-415-1419.

Sincerely,

/RA/

Leonard Olshan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure: As Stated

cc: See next page

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RISK-INFORMED INSPECTION NOTEBOOK FOR

ALVIN W. VOGTLE ELECTRIC GENERATING PLANT

UNITS 1 AND 2

PWR, WESTINGHOUSE, FOUR-LOOP PLANT WITH LARGE DRY CONTAINMENT

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Prepared for

U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Systems Analysis and Regulatory Effectiveness

NOTICE

This notebook was developed for the NRC's inspection teams to support risk-informed inspections. The activities involved in these inspections are discussed in "Reactor Oversight Process Improvement," SECY-99-007A, March 1999. The user of this notebook is assumed to be an inspector with an extensive understanding of plant-specific design features and operation. Therefore, the notebook is not a stand-alone document, and may not be suitable for use by non-specialists. This notebook will be periodically updated with new or replacement pages incorporating additional information on this plant. All recommendations for improvement of this document should be forwarded to the Chief, Probabilistic Safety Assessment Branch, NRR, with a copy to the Chief, Inspection Program Branch, NRR.

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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Alvin W. Vogtle Electric Generating Plant, Units 1 & 2.

The information includes the following: Categories of Initiating Events Table, Initiators and System Dependency Table, SDP Worksheets, and SDP Event Trees. This information is used by the NRC's inspectors to identify the significance of their findings, i.e., in screening risk-significant findings, consistent with Phase 2 screening in SECY-99-007A. The Categories of Initiating Event Table is used to determine the likelihood rating for the applicable initiating events. The SDP worksheets are used to assess the remaining mitigation capability rating for the applicable initiating event likelihood ratings in identifying the significance of the inspector's findings. The Initiators and System Dependency Table and the SDP Event Trees (the simplified event trees developed in preparing the SDP worksheets) provide additional information supporting the use of SDP worksheets.

The information contained herein is based on the licensee's Individual Plant Examination (IPE) submittal, the updated Probabilistic Risk Assessment (PRA), and system information obtained from the licensee during site visits as part of the review of earlier versions of this notebook. Approaches used to maintain consistency within the SDP, specifically within similar plant types, resulted in sacrificing some plant-specific modeling approaches and details. Such generic considerations, along with changes made in response to plant-specific comments, are summarized.

CONTENTS

TABLES

FIGURES

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1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

SECY-99-007A (NRC, March 1999) describes the process for making a Phase 2 evaluation of the inspection findings. The first step in this is to identify the pertinent core damage scenarios that require further evaluation consistent with the specifics of the inspection findings. To aid in this process, this notebook provides the following information:

1. Estimated Likelihood Rating for Initiating Event Categories
2. Initiators and System Dependency Table
3. Significance Determination Process (SDP) Worksheets
4. SDP Event Trees.

Table 1, Categories of Initiating Events, is used to estimate the likelihood rating for different initiating events for a given degraded condition and the associated exposure time at the plant. This Table follows the format of Table 1 in SECY-99-007A. Initiating events are grouped in frequency bins that are one order of magnitude apart. The Table includes the initiating events that should be considered for the plant and for which SDP worksheets are provided. The following initiating events are categorized by industry-average frequency: transients (Reactor Trip) (TRANS); transients without power conversion system (TPCS); large, medium, and small loss of coolant accidents (LLOCA, MLOCA, and SLOCA); inadvertent or stuck open relief valve (IORV or SORV); main steam line break (MSLB), anticipated transients without scram (ATWS), and interfacing system LOCA (ISLOCA). The frequency of the remaining initiating events vary significantly from plant to plant, and accordingly, they are categorized by plant-specific frequency obtained from the licensee. They include loss of offsite power (LOOP) and special initiators caused by loss of support systems.

The Initiators and System Dependency Table shows the major dependencies between frontline- and support-systems, and identifies their involvement in different types of initiators. This table identifies the most risk-significant systems; it is not an exhaustive nor comprehensive compilation of the dependency matrix, as known in Probabilistic Risk Assessments (PRAs). For pressurized water reactors (PWRs), the support systems/success criteria for Reactor Coolant Pump (RCP) seals are explicitly denoted to assure that the inspection findings on them are properly accounted for. This Table is used to identify the SDP worksheets to be evaluated, corresponding to the inspection's findings on systems and components.

To evaluate the impact of the inspection's findings on the core-damage scenarios, SDP worksheets are provided. There are two sets of SDP worksheets; one for those initiators that can be mitigated by redundant trains of safety systems, and the other for those initiators that cannot be mitigated; however, their occurrence is prevented by several levels of redundant barriers.

The first set of SDP worksheets contain two parts. The first identifies the functions, the systems, or combinations thereof that have mitigating functions, the number of trains in each system, and the number of trains required (success criteria) for the initiator. It also characterizes the mitigation capability in terms of the available hardware (e.g., 1 train, 1 multi-train system) and the operator action involved. The second part of the SDP worksheet contains the core-damage accident sequences associated with each initiator; these sequences are based on SDP event trees. In the parenthesis next to each sequence, the corresponding event-tree branch number(s) representing the sequence is given. Multiple branch numbers indicate that the different accident sequences identified by the event tree have been merged into one through Boolean reduction. The SDP worksheets are developed for each of the initiating event categories, including the "Special Initiators", the exception being those which directly lead to a core damage (the inspections of these initiators are assessed differently; see SECY-99-007A). The special initiators are those that are caused by complete or partial loss of support systems. A special initiator typically leads to a reactor scram and degrades some frontline or support systems (e.g., Loss of CCW in PWRs).

In considering the special initiators, we defined a set of criteria for including them to maintain some consistency across the plants. These conditions are as follows:

1. The special initiator should degrade at least one of the mitigating safety functions thereby changing its mitigation capability in the worksheet. For example, when a safety function with two redundant trains, classified as a multi-train system, degrades to a one-train system, it is classified as 1 Train, due to the loss of one of the trains as a result of the special initiator.

2. The special initiators which degrade the mitigation capability of the systems/functions associated with the initiator from comparable transient sequences by two and higher orders of magnitude must be considered.

From the above considerations, the following classes of initiators are considered in this notebook:

1. Transients with power conversion system (PCS) available, called Transients (Reactor trip) (TRANS),
2. Transients without PCS available, called Transients w/o PCS (TPCS),
3. Small Loss of Coolant Accident (SLOCA),
4. Stuck-open Power Operated Relief Valve (SORV),
5. Medium LOCA (MLOCA),
6. Large LOCA (LLOCA),
7. Steam Generator Tube Rupture (SGTR),
8. Anticipated Transients Without Scram (ATWS), and
9. Main Steam Line Break (MSLB).

Examples of special initiators included in the notebook are as follows:

1. Loss of Offsite Power (LOOP),
2. LOOP with failure of 1 Emergency AC bus or associated EDG (LEAC),
3. Loss of 1 DC Bus (LDC),

4. Loss of component cooling water (LCCW),
5. Loss of instrument air (LIA),
6. Loss of service water (LSW).

The worksheet for the LOOP includes LOOP with emergency AC power (EAC) available and LOOP without EAC, i.e., Station Blackout (SBO). LOOP with partial availability of EAC, i.e., LOOP with loss of a bus of EAC, is covered in a separate worksheet to avoid making the LOOP worksheet too large. In some plants, LOOP with failure of 1 EAC bus is a large contributor to the plant's core damage frequency (CDF).

The second set of SDP worksheets addresses those initiators that cannot be mitigated, i.e., can directly lead to core-damage. It currently includes the Interfacing System LOCA (ISLOCA) initiator. ISLOCAs are those initiators that could result in a loss of RCS inventory outside the containment, sometimes referred to as a "V" sequence. In PWRs, this event effectively bypasses the capability to utilize the containment sump recirculation once the RWST has emptied. Also, through bypassing the containment, the radiological consequences may be significant. In PWRs, this typically includes loss of RCS inventory through high- and low-pressure interfaces, such as RHR connections, RCP thermal barrier heat-exchanger, high-pressure injection piping if the design pressure (pump head) is much lower than RCS pressure, and, potentially, through excess letdown heat exchanger. RCS inventory loss through ISLOCA could vary significantly depending on the size of the leak path; some may be recoverable with minimal impact. The SDP worksheet for ISLOCA, therefore, identifies the major consequential leak paths, and the barriers that should fail, allowing the initiator to occur.

Following the SDP worksheets, the SDP event trees corresponding to each of the worksheets are presented. The SDP event trees are simplified event trees developed to define the accident sequences identified in the SDP worksheets. For special initiators whose event tree closely corresponds to another event tree (typically, the Transient (Reactor trip) or Transients w/o PCS event tree) with one or more functions eliminated or degraded, a separate event tree may not be drawn.

The following items were considered in establishing the SDP event trees and the core-damage sequences in the SDP worksheets:

1. Event trees and sequences were developed such that the worksheet contains all the major accident sequences identified by the plant-specific IPEs/PRAs. The special initiators modeled for a plant is based on a review of the special initiators included in the plant IPE/PRA and the information provided by the licensee.

2. The event trees and sequences for each plant take into account the IPE/PRA models and event trees for all similar plants. For modeling the response to an initiating event, any major deviations in one plant from similar plants may be noted at the end of the worksheet.

3. The event trees and the sequences were designed to capture core-damage scenarios, without including containment-failure probabilities and consequences. Therefore, branches of event

trees that are developed only for the purpose of a Level II PRA analysis are not considered. The resulting sequences are merged, using Boolean logic.

4. The simplified event trees focus on classes of initiators, as defined above. In so doing, many separate event trees in the IPEs/PRAs often are represented by a single tree. For example, some IPEs/PRAs define four classes of LOCAs rather than the three classes considered here. The sizes of LOCAs for which high-pressure injection is not required are sometimes divided into two classes, the only difference between them being the need for reactor scram in the smaller break size. There may be some consolidation of transient event trees besides defining the special initiators following the criteria defined above.

5. Major actions by the operator during accident scenarios are credited using four categories of Human Error Probabilities (HEPs). They are termed operator action =1 (representing an error probability of 5E-2 to 0.5), operator action=2 (error probability of 5E-3 to 5E-2), operator action=3 (error probability of 5E-4 to 5E-3), and operator action=4 (error probability of 5E-5 to 5E-4). An human action is assigned to a category bin, based on a generic grouping of similar actions among a class of plants. This approach resulted in designation of some actions to a higher bin, even though the IPE/PRA HEP value may have been indicative of a lower category. In such cases, it is noted at the end of the worksheet. On the other hand, if the IPE/PRA HEP value suggests a higher category than that generically assumed, the HEP is assigned to a bin consistent with the IPE/PRA value in recognition of potential plant-specific design; a note is also given in these situations. Operator's actions belonging to category 4, i.e., operator action=4, may only be noted at the bottom of worksheet because, in those cases, equipment failures may have the dominating influence in determining the significance of the findings.

The four sections that follow include Categories for Initiating Events Table, Initiators and Dependency Table, SDP worksheets, and the SDP event trees for Alvin W. Vogtle Electric Generating Plant, Units 1 & 2.

1.1 INITIATING EVENT LIKELIHOOD RATINGS

Table 1 presents the applicable initiating events for this plant and their estimated likelihood ratings corresponding to the exposure time for degraded conditions. The initiating events are grouped into rows based on their frequency. As mentioned earlier, loss of offsite power (LOOP) and special initiators are assigned to rows using the plant-specific frequency obtained from individual licensees. For other initiating events, industry-average values are used.

Table 1 Categories of Initiating Events for Alvin W. Vogtle Electric Generating Plant, Units 1 & 2

Row	Approximate Frequency	Example Event Type	Estimated Likelihood Rating		
I	> 1 per 1-10 yr	Reactor Trip (TRANS), Loss of Power Conversion System (TPCS)	A	B	C
II	1 per 10-10^2 yr	Loss of offsite power (LOOP)	B	C	D
III	1 per 10^2 - 10^3 yr	SGTR, Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures (SLOCA), MSLB (outside containment), Loss of One 125 V DC Bus (LBDC)	C	D	E
IV	1 per 10^3 - 10^4 yr	Medium LOCA (MLOCA), Loss of NSCW (LNSCW), LOOP and Loss of One ESF 4.16 kV AC Bus (LOAC)	D	E	F
V	1 per 10^4 - 10^5 yr	Large LOCA (LLOCA)	E	F	G
VI	less than 1 per 10^5 yr	ATWS[1], ISLOCA	F	G	H
			> 30 days	**3-30 days**	**< 3 days**
			Exposure Time for Degraded Condition		

Note:

1. The SDP worksheets for ATWS core damage sequences assume that the ATWS is not recoverable by manual actuation of the reactor trip function. Thus, the ATWS frequency to be used by these worksheets must represent the ATWS condition that can only be mitigated by the systems shown in the worksheet (e.g., boration). Any inspection finding that represents a loss of capability for manual reactor trip for a postulated ATWS scenario should be evaluated by a risk analyst to consider the probability of a successful manual trip.

1.2 INITIATORS AND SYSTEM DEPENDENCY

Table 2 lists the systems included in the SDP worksheets, the major components in the systems, and the support system dependencies. The systems' involvements in different initiating events are noted in the last column.

Table 2 Initiators and System Dependency for Alvin W. Vogtle, Units 1 & 2

Affected Systems	Major Components	Support Systems	Initiating Event
Auxiliary Component Cooling Water (ACCW)	Two 100% capacity pumps	4.16 kV AC, DC, NSCW	SLOCA
Accumulators (ACC)	Four ACCs	None	SLOCA, SORV, MLOCA, LLOCA, LNSCW, LOAC
Auxiliary Feedwater System (AFW)	Two MDPs[1]	4.16 kV AC, 480 V AC (for valves), DC, ESFAS, HVAC	All except LLOCA
	One TDP[1, 2]	ESFAS, DC (control power, operator backup)	
Component Cooling Water (CCW)[3]	Two trains with three 50% pumps each	4.16 kV AC, DC, ESFAS, NSCW	All except ATWS
Condensate / MFW	Three 50% Condensate pumps	Non-1E AC, Non-1E DC, Circulating water for Condenser, TPCCW	TRANS
	Two 50% turbine-driven MFPs	Non-1E DC, IA, TPCW	
CVCS[3]	Two centrifugal charging pumps (CCP)	4.16 kV AC, 480 V AC (for valves), DC, ESFAS, NSCW	All except LLOCA
	Two boric acid transfer pumps		ATWS

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Electric Power Systems	4.16 kV AC Power System: Two Class 1E ESF buses	HVAC, DC	All
	Two EDGs.	DC (to start EDGs), DG HVAC, NSCW, Fuel Oil Transfer	LOOP, LOAC
	480 V AC Power System	4.16 kV AC	All
	120 V AC Distribution System	DC Power System	All
	DC Power System: Four ESF buses, battery chargers and batteries	HVAC, 480 V AC (without AC, battery capacity is 4 hrs.)	All
Engineered Safety Features Actuation System (ESFAS)	Two redundant logic trains	120 VAC, Control Room Ventilation	All
Essential Chilled Water (ECW)	Two 100% capacity trains	480 V AC, NSCW, ESFAS	All
Heating, Ventilating, and Air Conditioning System (HVAC)	Control Building ESF Electrical Equipment Room HVAC	480 V AC, DC, ECW	All
	Room coolers (1555 system)		
Instrument Air (IA)	Three air compressors per unit & one swing compressor shared by two units	4.16 kV AC, DC, TPCCW	TRANS, MSLB
Main Steam (MS)	One Atmospheric Relief Valve (ARV) per SG	480 V AC, DC (AC & DC are required to open ARVs; backup is a hydraulic pump)	SLOCA, SORV, MLOCA, SGTR, LNSCW, LOAC
	Five Safety Valves per SG	None	TRANS, TPCS
	Two MSIVs per SG	DC, IA	MSLB

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Nuclear Service Cooling Water (NSCW)	Two trains with three 50% pumps each (split system)	4.16 kV AC, DC, ESFAS	All
Pressurizer Pressure Relief	Two PORVs	DC	All except MLOCA, LLOCA
	Two Block Valves	480 V AC	All except MLOCA, LLOCA, LOAC
	Three Safety Valves	None	ATWS
Reactor Coolant Pumps (RCPs)	Seals	1 / 2 charging pumps for seal injection and 1/ 2 ACCW pumps for thermal barrier cooling	SLOCA
RHR/LPSI[3]	Two RHR/LPSI pumps and heat exchangers	4.16 kV AC, 480 V AC (for valves), DC, ESFAS, CCW, NSCW	All except ATWS
Safety Injection System (SI)[3]	Two trains, each with one pump	4.16 kV AC, 480 V AC (for valves), DC, ESFAS, NSCW	All except LLOCA, ATWS
Turbine Plant Closed Cooling Water (TPCCW)	Two 100% pumps	Non-1E AC, Non-1E DC, TPCW	TRANS, MSLB
Turbine Plant Cooling Water (TPCW)	Two 100% pumps	Non-1E AC, Non-1E DC	TRANS, MSLB

Notes:

(1) According to the IPE's "Support System Dependency Matrix" (page C-11), the pumps of the AFW system do not require cooling.

(2) According to the IPE's "Support System Dependency Matrix" (page C-11), the TDP of the AFW system does not require room ventilation (HVAC).

(3) During review of the draft SDP document with the licensee, the dependency of this system on HVAC was removed.

Table 2 (Continued)

(4) Plant internal event CDF (including internal floods) = 2.4E-5/year.

Vogtle - 11 - Rev. 0, June 6, 2001

1.3 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Alvin W. Vogtle Electric Generating Plant Units 1 and 2. The SDP worksheets are presented for the following initiating event categories:

1. Transients with PCS Available (Reactor Trip) (TRANS)
2. Transients with Loss of PCS (TPCS)
3. Small LOCA (SLOCA)
4. Stuck-open PORV (SORV)
5. Medium LOCA (MLOCA)
6. Large LOCA (LLOCA)
7. Loss of Offsite Power (LOOP)
8. Steam Generator Tube Rupture (SGTR)
9. Anticipated Transients without Scram (ATWS)
10. Main Steam Line Break Outside Containment (MSLB)
11. Loss of NSCW (LNSCW)
12. Loss of One 125 V DC Bus (LBDC)
13. LOOP and Loss of One ESF 4.16 kV AC Bus (LOAC)
14. Interfacing Systems LOCA (ISLOCA)

Table 3.1 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Transients with PCS Available (Reactor Trip) (TRANS)

Estimated Frequency (Table 1 Row) _________	**Exposure Time** _________	**Table 1 Result (circle): A B C D E F G H**	
Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**		
Power Conversion System (PCS)	1/2 Main Feedwater trains with 1/3 condensate trains (operator action = 2) (1)		
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs with 1/5 SG safety valves on each SG fed by AFW (2)		
Early Inv., High Pressure Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 SI pumps (1 multi-train system) to 3/4 cold legs		
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORVs and block valves open for Feed/Bleed (operator action = 2)		
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TRANS - PCS - AFW - HPR (4)			
2 TRANS - PCS - AFW - FB (5)			
3 TRANS - PCS - AFW - EIHP (6)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) When Main Feedwater pumps are not available but condensate pumps are available, the condensate pumps can be used for SG injection bypassing feed water systems.

(2) The success criteria in the IPE (page 3-19) does not include the use of ARVs.

Table 3.2 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Transients with Loss of PCS (TPCS)

Estimated Frequency (Table 1 Row) __________ **Exposure Time** __________ **Table 1 Result (circle): A B C D E F G H**

<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs with 1/5 SG safety valves on each SG fed by AFW
Early Inv., High Pressure Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 SI pumps (1 multi-train system) to 3/4 cold legs
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORVs and block valves open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	**<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>**	**<u>Sequence Color</u>**
1 TPCS - AFW - HPR (3)			
2 TPCS - AFW - FB (4)			
3 TPCS - AFW - EIHP (5)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.3 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Small LOCA (SLOCA)

Estimated Frequency (Table 1 Row) __________ **Exposure Time __________** **Table 1 Result (circle): A B C D E F G H**

<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
Early Inv., High Pressure Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 SI pumps (1 multi-train system) to 3/4 cold legs
RCS Cooldown / Depressurization (DEP1)[1]	Operator depressurizes RCS using 2/4 secondary ARVs [(2)] with 1/2 PORVs and block valves (operator action = 3)
RCS Cooldown / Depressurization (DEP2)[1]	Operator depressurizes RCS using 2/4 secondary ARVs [(2)] with 1/2 PORVs and block valves (operator action = 2)
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORVs and block valves open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/3 remaining accumulators [(3)] with 1/2 RHR trains to 2/4 cold legs (1 multi-train system)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs
Low Pressure Recirculation (LPR)	1/2 RHR pumps with operator action for switchover (operator action = 3) to 2/4 cold legs

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	**<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>**	**<u>Sequence Color</u>**
1 SLOCA - LPR (2, 6)			
2 SLOCA - DEP1 - HPR (4)			
3 SLOCA - EIHP - LPI (7)			

4 SLOCA - EIHP - DEP2 (8)			
5 SLOCA - AFW - HPR (10)			
6 SLOCA - AFW - FB (11)			
7 SLOCA - AFW - EIHP (12)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

Notes:

(1) The licensee distinguishes two types of RCS Cooldown / Depressurization: when high pressure injection is available (normal cooldown), and when it is not available (rapid cooldown for LPI). The licensee uses the same equipment and success criteria for both types of RCS Cooldown / Depressurization, but different human error probabilities (HEPs). For normal cooldown, the licensee assigns HEP = 3.87E-3, and for rapid cooldown for LPI, the licensee assigns HEP = 7.74E-3.

(2) The licensee also credits 3/3 turbine bypass valves (TBVs) for RCS Cooldown / Depressurization. It appears that this mode of depressurization requires the use of the Condenser Steam Dump Valves. Since the equipment involved and success criteria are not clear, we do not give credit to the TBVs at this time.

(3) Accumulators are needed for inventory makeup during depressurization for low pressure injection when HPI fails. We assume that one of the accumulators is unavailable due to the break.

Table 3.4 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Stuck Open PORV (SORV)[1]

Estimated Frequency (Table 1 Row)	Exposure Time	Table 1 Result (circle): A B C D E F G H

<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**
Isolation of Small LOCA (BLK)	Automatic signal closes the block valve associated with stuck open PORV (1 train) [2]
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
Early Inv., High Pressure Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 SI pumps (1 multi-train system) to 3/4 cold legs
RCS Cooldown / Depressurization (DEP1)[3]	Operator depressurizes RCS using 2/4 secondary ARVs [4] with 1/2 PORVs and block valves (operator action = 3)
RCS Cooldown / Depressurization (DEP2)[3]	Operator depressurizes RCS using 2/4 secondary ARVs [4] with 1/2 PORVs and block valves (operator action = 2)
Primary Heat Removal, Feed/Bleed (FB)	Operator carries out Feed/Bleed (operator action = 2) [5]
Low Pressure Injection (LPI)	1/3 remaining accumulators [6] with 1/2 RHR trains to 2/4 cold legs (1 multi-train system)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs
Low Pressure Recirculation (LPR)	1/2 RHR pumps with operator action for switchover (operator action = 3) to 2/4 cold legs

<u>Circle Affected Functions</u>	<u>Recovery of Failed Train</u>	<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>	<u>Sequence Color</u>
1 SORV - BLK - LPR (2, 6)			
2 SORV - BLK - DEP1 - HPR (4)			
3 SORV - BLK - EIHP - LPI (7)			
4 SORV - BLK - EIHP - DEP2 (8)			

5 SORV - BLK - AFW - HPR (10)			
6 SORV - BLK - AFW - FB (11)			
7 SORV - BLK - AFW - EIHP (12)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The sequences of the SDP SORV worksheet are the same as those of the SDP small LOCA event tree with the addition of the failure of the safety function "Isolation of Small LOCA (BLK)" after the initiating event, SORV.

(2) The licensee indicated that closing the PORV block valve to isolate the stuck open PORV is automatic, that is, there is an automatic signal.

(3) The licensee distinguishes two types of RCS Cooldown / Depressurization: when high pressure injection is available (normal cooldown), and when it is not available (rapid cooldown for LPI). The licensee uses the same equipment and success criteria for both types of RCS Cooldown / Depressurization, but different human error probabilities (HEPs). For normal cooldown, the licensee assigns HEP = 3.87E-3, and for rapid cooldown for LPI, the licensee assigns HEP = 7.74E-3.

(4) The licensee also credits 3/3 turbine bypass valves (TBVs) for RCS Cooldown / Depressurization. It appears that this mode of depressurization requires the use of the Condenser Steam Dump Valves. Since the equipment involved and success criteria are not clear, we do not give credit to the TBVs at this time.

(5) The stuck open PORV provides the bleed path for Feed/Bleed.

(6) Accumulators are needed for inventory makeup during depressurization for low pressure injection when HPI fails. We assume that one of the accumulators is unavailable due to the break.

Table 3.5 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Medium LOCA (MLOCA)

Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H

<u>**Safety Functions Needed**</u>**:**	<u>**Full Creditable Mitigation Capability for Each Safety Function**</u>**:**
Early Inventory, HP Injection (EIHP)	2/4 charging / safety injection pumps (1 multi-train system) to 2/3 intact cold legs
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
RCS Cooldown/Depressurization (DEPR)	Operator depressurizes RCS using 2/4 secondary ARVs (operator action = 2) (1)
Accumulators (ACC)	2/3 remaining accumulators (1 multi-train system)
Low Pressure Injection (LPI)	1/2 RHR pumps to 2/3 intact cold legs (1 multi-train system) (2)
High Pressure Recirculation (HPR)	2/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 2/3 cold legs
Low Pressure Recirculation (LPR)	1/2 RHR pumps with operator action for switchover (operator action = 3) to 1/3 cold legs

<u>**Circle Affected Functions**</u>	<u>**Recovery of Failed Train**</u>	<u>**Remaining Mitigation Capability Rating for Each Affected Sequence**</u>	<u>**Sequence Color**</u>
1 MLOCA - LPR (2, 9)			
2 MLOCA - LPI (3, 10)			
3 MLOCA - DEPR - HPR (5)			
4 MLOCA - AFW - HPR (7)			
5 MLOCA - EIHP - ACC (11)			

6 MLOCA - EIHP - DEPR (12)			
7 MLOCA - EIHP - AFW (13)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

Notes:

(1) The licensee distinguishes two types of RCS Cooldown / Depressurization: when high pressure injection is available (normal cooldown), and when it is not available (rapid cooldown for LPI). The licensee uses the same equipment and success criteria for both types of RCS Cooldown / Depressurization, but different human error probabilities (HEPs). For normal cooldown, the licensee assigns HEP = 1.72E-2, and for rapid cooldown for LPI, the licensee assigns HEP = 3.44E-2. Accordingly, we assigned a credit = 2.

(2) Human error probability (HEP) = 8.6E-4, limited by hardware failure.

Table 3.6 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Large LOCA (LLOCA)

Estimated Frequency (Table 1 Row) __________ **Exposure Time __________** **Table 1 Result (circle): A B C D E F G H**

<u>**Safety Functions Needed**</u>:	<u>**Full Creditable Mitigation Capability for Each Safety Function**</u>:
Accumulators (ACC)	3/3 remaining accumulators (1 train)
Low Pressure Injection (LPI)	1/2 RHR pumps to 2/3 intact cold legs (1 multi-train system)
Low Pressure Recirculation (LPR)	1/2 RHR pumps with operator action for switchover (operator action = 3) to 1/3 cold legs

<u>**Circle Affected Functions**</u>	<u>**Recovery of Failed Train**</u>	<u>**Remaining Mitigation Capability Rating for Each Affected Sequence**</u>	<u>**Sequence Color**</u>
1 LLOCA - LPR (2)			
2 LLOCA - LPI (3)			
3 LLOCA - ACC (4)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Table 3.7 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Loss of Offsite Power (LOOP)

Estimated Frequency (Table 1 Row) __________ **Exposure Time** __________ **Table 1 Result (circle): A B C D E F G H**

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Emergency AC Power (EAC)	1/2 Emergency Diesel Generators (1 multi-train system)
Turbine-driven AFW pump (TDAFW)	1/1 TDP trains of AFW to 2/4 SGs (1 ASD train)
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train to 2/4 SGs (1 ASD train)
Recovery of AC Power in < 1 hrs (REC1)	AC power recovered (operator action = 1) [1]
Recovery of AC Power in < 5 hrs (REC5)	AC power recovered (operator action = 2) [2]
Early Inventory, HP Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 safety injection pumps (1 multi-train system) to 3/4 cold legs
Early Inventory, HP Injection (CCP)	2/2 charging pumps [3] to 3/4 cold legs (1 train)
Primary Heat Removal (FB)	Operator uses RCS pressurizer 1/2 PORVs and block valves (operator action = 2)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - AFW - HPR (3)			
2 LOOP - AFW - FB (4)			
3 LOOP - AFW - EIHP (5)			
4 LOOP - EAC - HPR (7, 11) (AC recovered)			

5 LOOP - EAC - EIHP (8) (AC recovered)			
6 LOOP - EAC - REC5 (9)			
7 LOOP - EAC - TDAFW - FB (12) (AC recovered)			
8 LOOP - EAC - TDAFW - CCP (13) (AC recovered)			
9 LOOP - EAC - TDAFW - REC1 (14)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The probabilities of AC power not recovered are presented in pages 36 and 37 of response to RAI.

(2) In an SBO situation, an RCP seal LOCA may occur, with subsequent core damage at about 5 hours.

(3) The licensee does not credit the use of the safety injection pumps for feed and bleed in the sequences where EAC failed and AC power is later restored. In this scenario, 2 / 2 charging pumps are required.

Table 3.8 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Steam Generator Tube Rupture (SGTR)[1]

Estimated Frequency (Table 1 Row) __________ **Exposure Time __________** **Table 1 Result (circle): A B C D E F G H**

<u>**Safety Functions Needed**</u>:	<u>**Full Creditable Mitigation Capability for Each Safety Function**</u>:
Secondary Heat Removal (AFW)	1/2 MDPs of AFW (1 multi-train system) or 1/1 TDP of AFW (1 ASD Train) to 3/3 intact SGs
Early Inventory, HP Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 safety injection pumps (1 multi-train system) to 3/4 cold legs
Isolation of the Ruptured SG (ISOL)	Operator isolates the affected SG (operator action = 2)
Pressure Equalization (EQ)	Operator depressurizes RCS using 2/3 SG ARVs to less than setpoint of relief valves of SG (operator action = 2)
Feed-and-Bleed (FB)	Operator uses 1/2 RCS pressurizer PORVs and block valves (operator action = 2)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with 1/2 RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs

<u>**Circle Affected Functions**</u>	<u>**Recovery of Failed Train**</u>	<u>**Remaining Mitigation Capability Rating for Each Affected Sequence**</u>	<u>**Sequence Color**</u>
1 SGTR - EIHP - EQ (3)			
2 SGTR - EIHP - ISOL (4)			
3 SGTR - AFW - HPR (6)			
4 SGTR - AFW - FB (7)			
5 SGTR - AFW - EIHP (8)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Note:

(1) When secondary heat removal and high pressure injection are available, the licensee considers that failure of "Isolation of the Ruptured SG (ISOL)" or failure of "Pressure Equalization (EQ)" does not lead to core damage, even though it increases radioactive release outside containment. The RWST has a total tank capacity of 715,000 gallons with a minimum allowable volume of 631,478 gallons. The licensee considers that if high pressure injection fails, failure of ISOL or EQ leads to core damage.

Table 3.9 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Anticipated Transients without Scram (ATWS)

Estimated Frequency (Table 1 Row) __________	**Exposure Time** __________	**Table 1 Result (circle): A B C D E F G H**	
Safety Functions Needed: **Turbine trip (TTP)** **Secondary Heat Removal (AFW)** **Primary Relief (SRV)** **Emergency Boration (EMBO)**	**Full Creditable Mitigation Capability for Each Safety Function:** Automatic trip by AMSAC (1 train) 3/3 AFW pumps (1 ASD train) to 4/4 SGs 3/3 SRVs with 2/2 PORVs with associated block valves open (1 train) Operator conducts emergency boration using 1/2 charging pumps with 1/2 boric acid transfer pumps (operator action = 1) (1)		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 ATWS - EMBO (2)			
2 ATWS - SRV (3)			
3 ATWS - AFW (4)			
4 ATWS - TTP (5)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The human error probability (HEP) assessed by the licensee for emergency boration is 1.55E-1.

Table 3.10 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Main Steam Line Break Outside Containment (MSLB)

Estimated Frequency (Table 1 Row) __________ **Exposure Time** __________ **Table 1 Result (circle): A B C D E F G H**

Safety Functions Needed:	Full Creditable Mitigation Capability for Each Safety Function:
MSLB Isolated (MSIV)[1]	1/2 MSIVs close in 3/4 steam paths [2] (1 multi-train system)
Early Inventory, HP Injection (EIHP)	1/2 charging pumps (1 multi-train system) or 1/2 safety injection pumps (1 multi-train system) to 3/4 cold legs
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
Feedwater valves close (FWVC)	Operators close the valves feeding the SG whose MSIV did not close (operator action = 1) [3]
Stop Injection (STIN)	Operators stop high pressure injection (operator action = 2) [4]
Primary Heat Removal, Feed/Bleed (FB)	1/2 pressurizer PORVs with associated block valves open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/4 charging pumps/safety injection pumps with ½ RHR pumps and with operator action for switchover (operator action = 3) to 3/4 cold legs

Circle Affected Functions	Recovery of Failed Train	Remaining Mitigation Capability Rating for Each Affected Sequence	Sequence Color
1 MSLB - FWVC - STIN (3)			
2 MSLB - AFW - HPR (5)			
3 MSLB - AFW - FB (6)			

4 MSLB - EIHP - FWVC (8)			
5 MSLB - EIHP - AFW (9)			
6 MSLB - MSIV (10)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

1. When safety function MSIV fails, a major concern is pressurized thermal shock (PTS). We assumed it leads to core damage.
2. There are 2 MSIVs in series in each main steam line. The success criteria for “MSLB Isolated (MSIV)” requires the closure of at least 1 MSIV in each of three steam lines.
3. Since this action would be carried out under time and stress conditions, we assigned a mitigating credit = 1.
4. Operators stop high pressure injection to prevent pressurized thermal shock. Since this action would be carried out under time and stress conditions, but is expected to be relatively simple, we assigned a mitigating credit = 2.

Table 3.11 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Loss of NSCW (LNSCW)[1]

Estimated Frequency (Table 1 Row) _________	Exposure Time _________	Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Recovery of Cooling to One Train of Mitigating Equipment (REC1)	Plant staff recovers one NSCW pump to provide cooling to one train of mitigating equipment (operator action = 2)
Secondary Heat Removal (AFW)	1/1 MDAFW train (1 train) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
Early Inv., High Pressure Injection (EIHP)	1/1 charging pump (1 train) or 1/1 SI pump (1 train) to 3/4 cold legs
RCS Cooldown / Depressurization (DEP1)[2]	Operator depressurizes RCS using 2/4 secondary ARVs [3] with 1/2 PORVs and block valves (operator action = 3)
RCS Cooldown / Depressurization (DEP2)[2]	Operator depressurizes RCS using 2/4 secondary ARVs [3] with 1/2 PORVs and block valves (operator action = 2)
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORVs and block valves open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/4 accumulators [4] with 1/1 RHR train to 2/4 cold legs (1 train)
High Pressure Recirculation (HPR)	1/2 charging pumps/safety injection pumps with 1/1 RHR pump and with operator action for switchover (1 train) [6] to 3/4 cold legs
Low Pressure Recirculation (LPR)	1/1 RHR pump with operator action for switchover (1 train) [6] to 2/4 cold legs

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LNSCW - LPR (2, 6)			
2 LNSCW - DEP1 - HPR (4)			
3 LNSCW - EIHP - LPI (7)			
4 LNSCW - EIHP - DEP2 (8)			

5 LNSCW - AFW - HPR (10)			
6 LNSCW - AFW - FB (11)			
7 LNSCW - AFW - EIHP (12)			
8 LNSCW - REC1 (13)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

Notes:

1. The licensee provided useful comments on the event trees included in the previous version of the SDP document, but it did not provide information on the event trees related to failure of support systems. This discussion on loss of NSCW is based on the information given on the IPE's "Initiator to System Dependency Matrix" (page C-3), and on note 24 of page C-6. The NSCW provides cooling for the following components: EDGs, CCW heat exchangers, ACCW heat exchangers, ECW chillers, and ECCS (RHR, SIP, CCP) pump motor coolers and CS pump motor coolers. ACCW can continue for a short time (approximately half an hour) before system failure occurs. Since the motor-driven pumps of AFW depend on HVAC, and HVAC depends on ECW, we assume that one of the pumps will be unavailable. The IPE (pages 3-126 and 3-129) credits "establish one pump NSCW operation", with a HEP = 3.93E-3. However, since we expect that this action will be carried out under time and stress constraints, we assigned a credit = 2. We assume that this restoration provides cooling to one train of mitigating equipment, even though each NSCW pump has 50% capacity. We assume that this recovery action occurs after an RCP seal LOCA has occurred, but before core damage. A non-recoverable loss of NSCW causes an unmitigated RCP seal LOCA. The licensee does not credit the use of PCS in a small LOCA, so we do not give credit to PCS in a loss of NSCW. The frequency of loss of NSCW is 1.4E-4 / year (IPE, page 3-6). A sequence with the initiating event loss

of NSCW was not found in the 100 dominant accident sequences presented in the IPE (Table 3.4-3). Loss of NSCW contributes 0.05% to the total CDF.

2. The licensee distinguishes two types of RCS Cooldown / Depressurization: when high pressure injection is available (normal cooldown), and when it is not available (rapid cooldown for LPI). The licensee uses the same equipment and success criteria for both types of RCS Cooldown / Depressurization, but different human error probabilities (HEPs). For normal cooldown, the licensee assigns HEP = 3.87E-3, and for rapid cooldown for LPI, the licensee assigns HEP = 7.74E-3.

3. The licensee also credits 3/3 turbine bypass valves (TBVs) for RCS Cooldown / Depressurization. It appears that this mode of depressurization requires the use of the Condenser Steam Dump Valves. Since the equipment involved and success criteria are not clear, we do not give credit to the TBVs at this time.

4. Accumulators are needed for inventory makeup during depressurization for low pressure injection when HPI fails.

5. Operator action = 2, limited by hardware failure.

6. Operator action = 3, limited by hardware failure.

Table 3.12 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Loss of One 125 V DC Bus (LBDC)[1]

Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function**:
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 train) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs
Early Inv., High Pressure Injection (EIHP)	1/1 charging pump (1 train) or 1/1 SI pump (1 train) to 3/4 cold legs
Primary Heat Removal, Feed/Bleed (FB)	1/1 PORV and block valve open for Feed/Bleed (1 train) (2)
High Pressure Recirculation (HPR)	1/2 charging pumps/safety injection pumps with 1/1 RHR pumps and with operator action for switchover (1 train) (3) to 3/4 cold legs

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LBDC - AFW - HPR (3)			
2 LBDC - AFW - FB (4)			
3 LBDC - AFW - EIHP (5)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

1. There are four 125 V DC ESF buses per unit. The IPE selected loss of 125 V DC bus 1AD1 and loss of 125 V DC bus 1BD1 as initiating events. The licensee provided useful comments on the event trees included in the previous version of the SDP document, but it did not provide information on the event trees related to failure of support systems. This discussion on loss of one 125 V DC bus is based on the information given on the IPE's "Initiator to System Dependency Matrix" (page C-3), and on note 26 of page C-6. This note discusses the loss of 125 V DC bus 1BD1; it appears that the loss of 125 V DC bus 1AD1 would have a symmetrical effect. On a loss of 125 V DC bus 1BD1, main feedwater isolation, isolation bypass, main flow control, and startup flow control valves close resulting in feedwater isolation; main steam isolation, and isolation bypass train A valves close, resulting in steamline isolation; and reactor and turbine trip occurs from loss of main feedwater. Only the train of safeguards equipment aligned to the opposite DC bus would be operable. Because the MSIVs have closed, steam dump to condenser is not available. The operator must use local manual control to open two of the main steam ARVs because the solenoids for these valves have fail closed. One pressurizer PORV is inoperable. The frequency of loss of 125 V DC bus 1BD1 or 1AD1 is 1.8E-3 / year. The event tree for loss of 125 V DC bus 1BD1 (1AD1) is the same as the one for Transients with Loss of PCS (TPCS).

2. Operator action = 2, limited by hardware failure.

3. Operator action = 3, limited by hardware failure.

Table 3.13 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — LOOP and Loss of One ESF 4.16 kV AC Bus (LOAC)[(1)]

Estimated Frequency (Table 1 Row) _________	**Exposure Time _________**		**Table 1 Result (circle): A B C D E F G H**
Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**		
PORV Recloses (PORV)	2/2 pressurizer PORVs reclose after opening during transient (1 train) [(2)]		
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 train) or 1/1 TDAFW train (1 ASD train) to 2/4 SGs		
Early Inv., High Pressure Injection (EIHP)	1/1 charging pump (1 train) or 1/1 SI pump (1 train) to 3/4 cold legs		
RCS Cooldown / Depressurization (DEP1)[3]	Operator depressurizes RCS using 2/4 secondary ARVs [(4, 5)] (operator action = 3)		
RCS Cooldown / Depressurization (DEP2)[3]	Operator depressurizes RCS using 2/4 secondary ARVs [(4, 5)] (operator action = 2)		
Primary Heat Removal, Feed/Bleed (FB1)	1/2 PORVs and block valves open for Feed/Bleed (operator action = 2)		
Primary Heat Removal, Feed/Bleed (FB2)	Operator carries out Feed/Bleed [(6)] (operator action = 2)		
Low Pressure Injection (LPI)	1/3 remaining accumulators [(7)] with 1/1 RHR train to 2/4 cold legs (1 train)		
High Pressure Recirculation (HPR)	1/2 charging pumps/safety injection pumps with 1/1 RHR pump and with operator action for switchover (1 train) [(8)] to 3/4 cold legs		
Low Pressure Recirculation (LPR)	1/1 RHR pump with operator action for switchover (1 train) [(8)] to 2/4 cold legs		

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOAC - AFW - HPR (3, 15)			
2 LOAC - AFW - FB1 (4)			
3 LOAC - AFW - EIHP (5, 17)			
4 LOAC - PORV - LPR (7, 11)			

5 LOAC - PORV - DEP1 - HPR (9)			
6 LOAC - PORV - EIHP - LPI (12)			
7 LOAC - PORV - EIHP - DEP2 (13)			
8 LOAC - PORV - AFW - FB2 (16)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

<u>Notes</u>:

1. There are two Class 1E ESF 4.16 kV buses per unit. Each bus supports one train of safeguards equipment (IPE, page C-12, notes 1 and 2), so loss of one ESF 4.16 kV bus would cause the loss of one train of safeguards equipment.

2. The motive power for the PORV block valves is 480 VAC power. Hence, if one ESF 4.16 kV bus is lost, there may not be motive power available to close the block valve of a stuck open PORV.

3. The licensee distinguishes two types of RCS Cooldown / Depressurization: when high pressure injection is available (normal cooldown), and when it is not available (rapid cooldown for LPI). The licensee uses the same equipment and success criteria for both types of RCS Cooldown / Depressurization, but different human error probabilities (HEPs). For normal cooldown, the licensee assigns HEP = 3.87E-3, and for rapid cooldown for LPI, the licensee assigns HEP = 7.74E-3.

4. The licensee also credits 3/3 TBVs for RCS Cooldown / Depressurization. It appears that TBV stands for turbine bypass valve, and that this mode of depressurization requires the use of the Condenser Steam Dump Valves. Since the equipment involved and success criteria are not clear, we do not give credit to this equipment at this time.

5. Depressurization also requires 1 / 2 PORVs and block valves. This requirement is provided by the stuck open PORV.

6. Feed/Bleed requires 1 / 2 PORVs and block valves open. This requirement is provided by the stuck open PORV.

7. Accumulators are needed for inventory makeup during depressurization for low pressure injection when HPI fails.

8. Operator action = 3, limited by hardware failure.

Table 3.14 SDP Worksheet for Alvin W. Vogtle Nuclear Plant, Units 1 and 2 — Interfacing Systems LOCA (ISLOCA)[1]

<table>
<tr><td colspan="4">Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H</td></tr>
<tr><td><u>Safety Functions Needed</u>:

RHR Hot Leg Suction Line
Low Pressure Injection System [3]
High Pressure Injection System (suction) [3]
Chemical and Volume Control System (suction) [3]
Component Cooling Water System (thermal barrier heat exchanger) [3]</td><td colspan="3"><u>Full Creditable Mitigation Capability for Each Safety Function</u>:

Failure of both motor operated valves [2].</td></tr>
<tr><td><u>Circle Affected Functions</u></td><td><u>Recovery of Failed Train</u></td><td><u>Remaining Mitigation Capability Rating for Each Affected Sequence</u></td><td><u>Sequence Color</u></td></tr>
<tr><td colspan="4">Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.</td></tr>
</table>

Notes:

1. This worksheet is different from the other worksheets in that ISLOCA is typically an unmitigated initiating event in most PRAs. Therefore, the right side of the worksheet contains paths which may lead to an ISLOCA rather than mitigating systems to address an event in progress. As such, it is not intended to be referenced by the last column of Table 2, Initiators and System Dependency Table.
2. From IPE, page 3-25.
3. These are systems that typically have a high- low pressure interface such that failure of the barriers of the interface would lead to a LOCA outside the containment. They are included here for completeness.

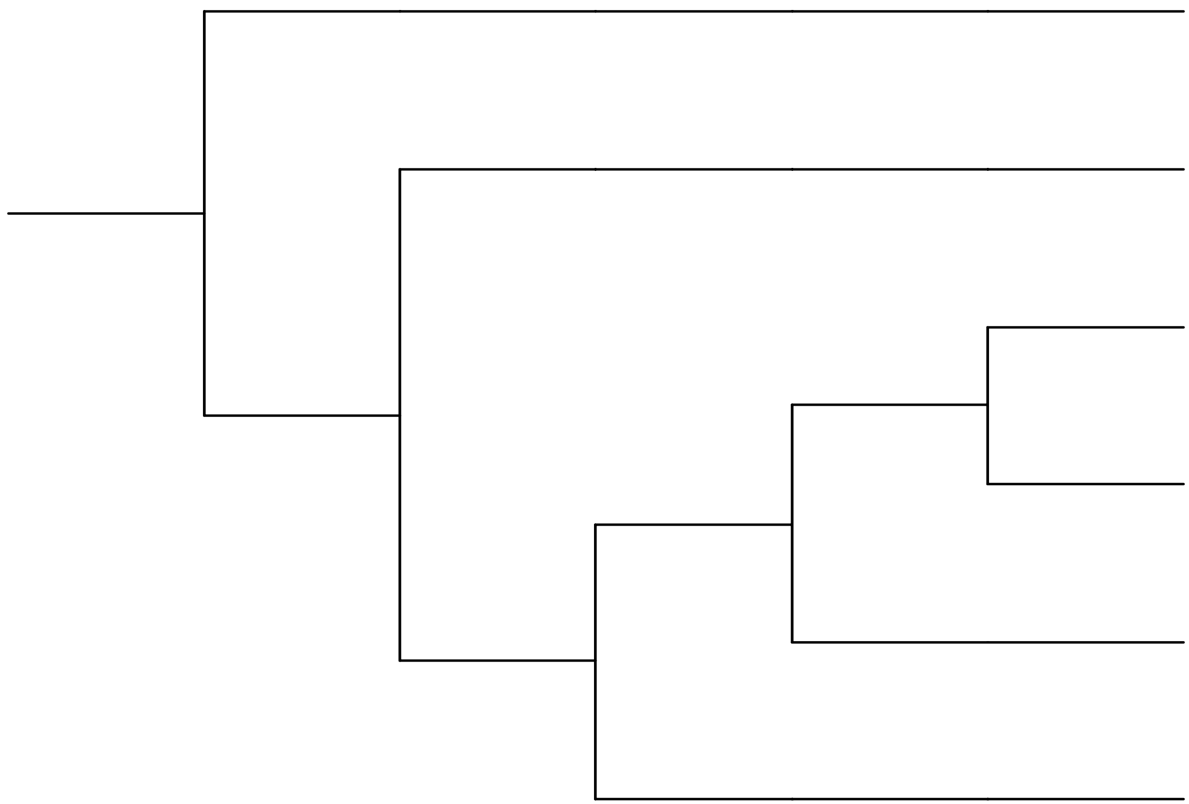
1.4 SDP Event Trees

This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined in the corresponding SDP worksheets.

The following event trees are included:

1. Transients with PCS Available (Reactor Trip) (TRANS)
2. Transients with Loss of PCS (TPCS)
3. Small LOCA (SLOCA)
4. Medium LOCA (MLOCA)
5. Large LOCA (LLOCA)
6. Loss of Offsite Power (LOOP)
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients without Scram (ATWS)
9. Main Steam Line Break Outside Containment (MSLB)
10. Loss of NSCW (LNSCW)
11. LOOP and Loss of One ESF 4.16 kV AC Bus (LOAC)

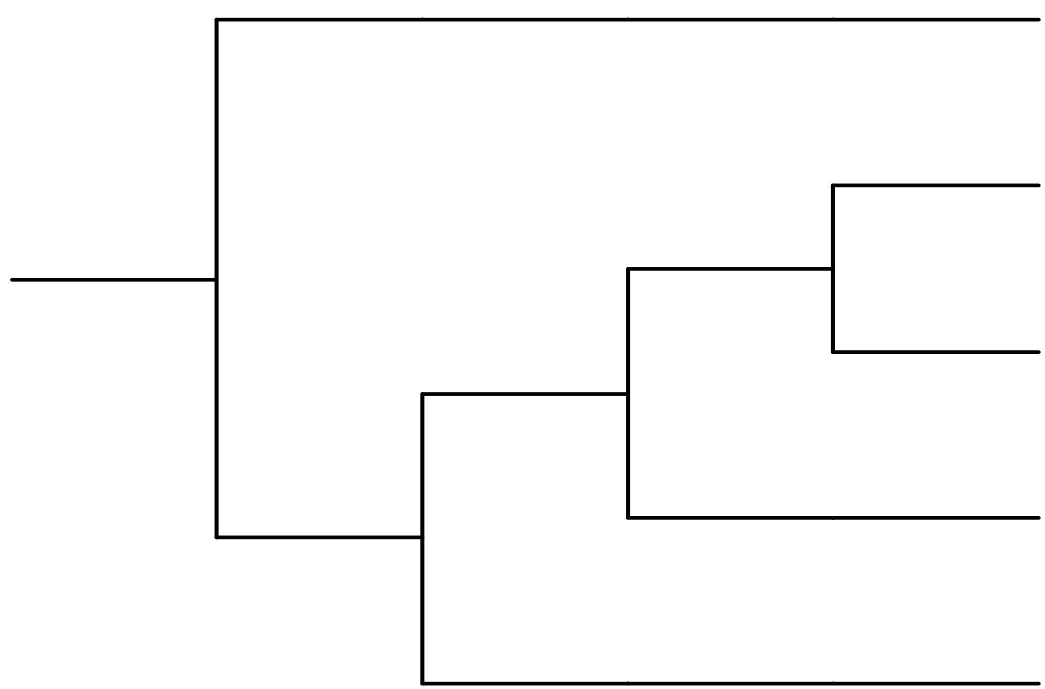
TRANS	PCS	AFW	EIHP	FB	HPR	#	STATUS



#	STATUS
1	OK
2	OK
3	OK
4	CD
5	CD
6	CD

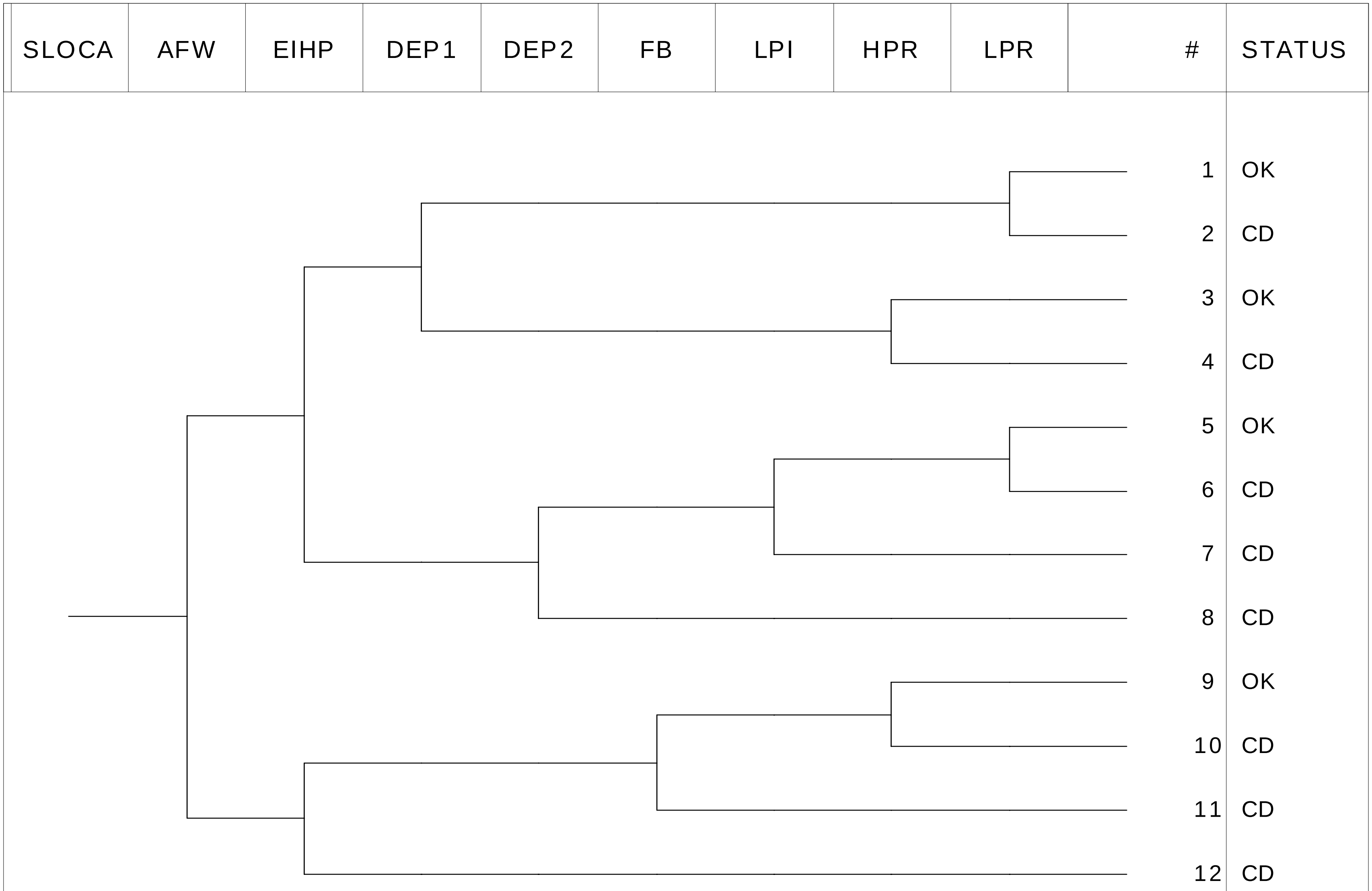
Plant Name Abbrev.:VOGT

TPCS	AFW	EIHP	FB	HPR	#	STATUS

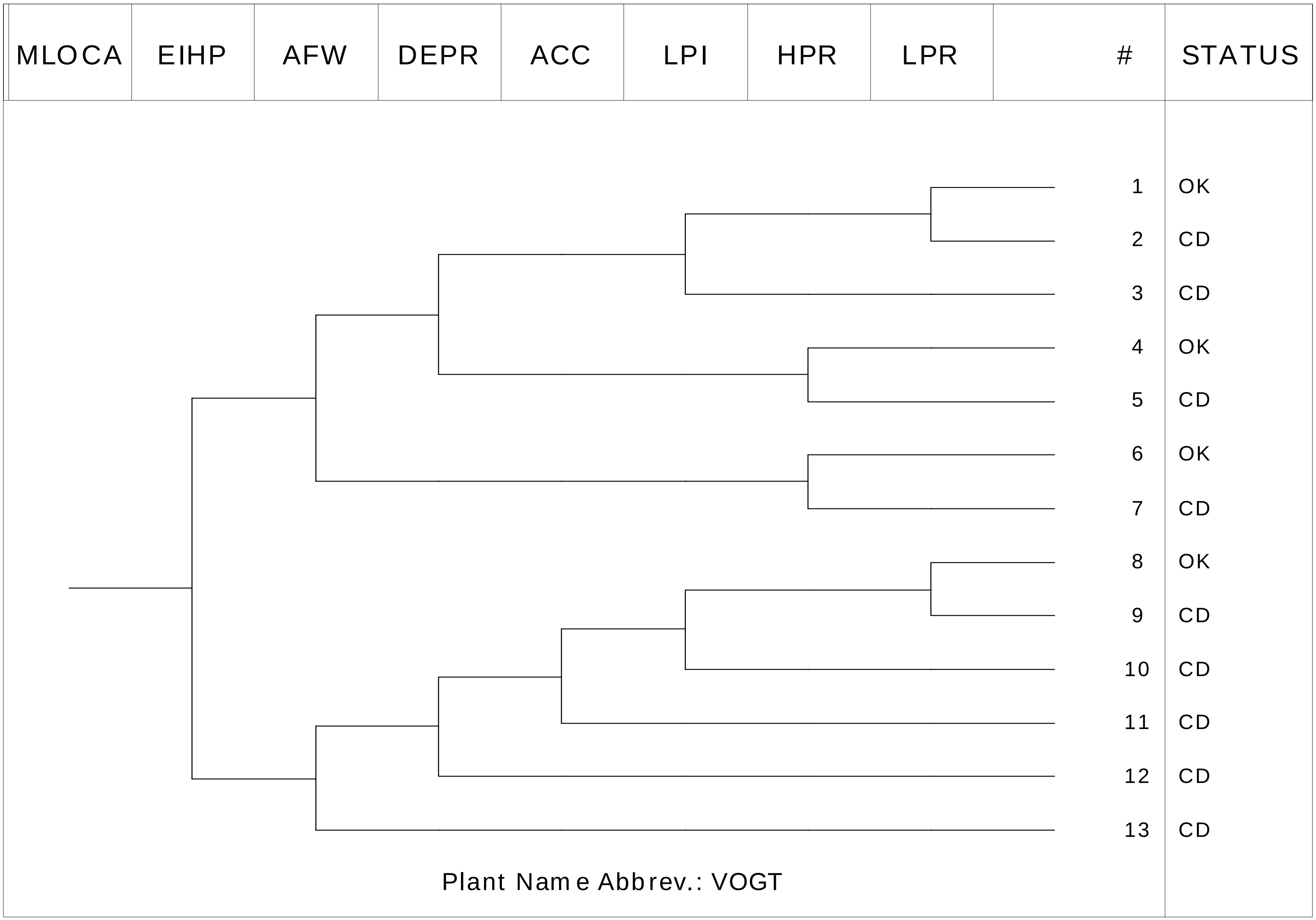


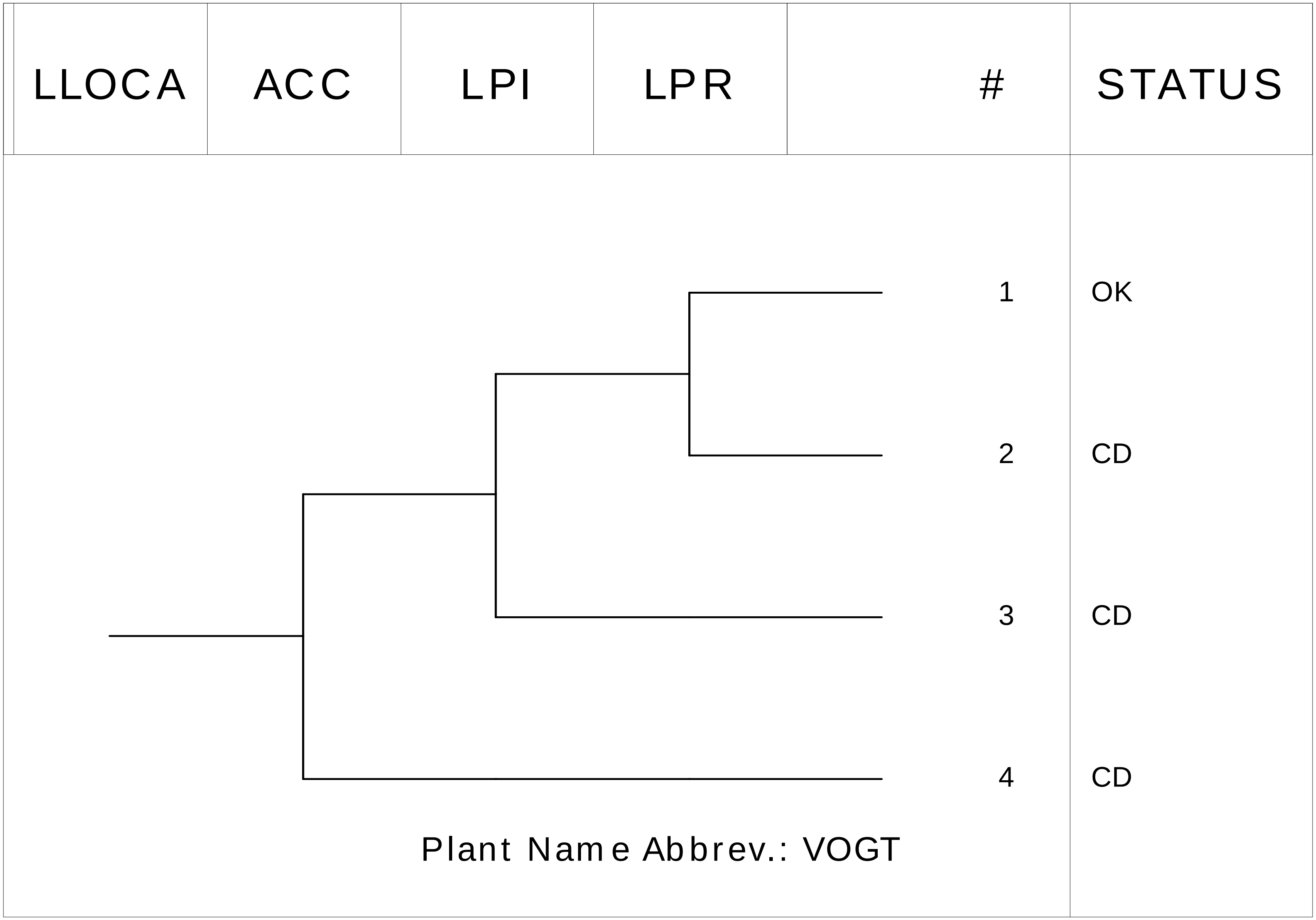
#	STATUS
1	OK
2	OK
3	CD
4	CD
5	CD

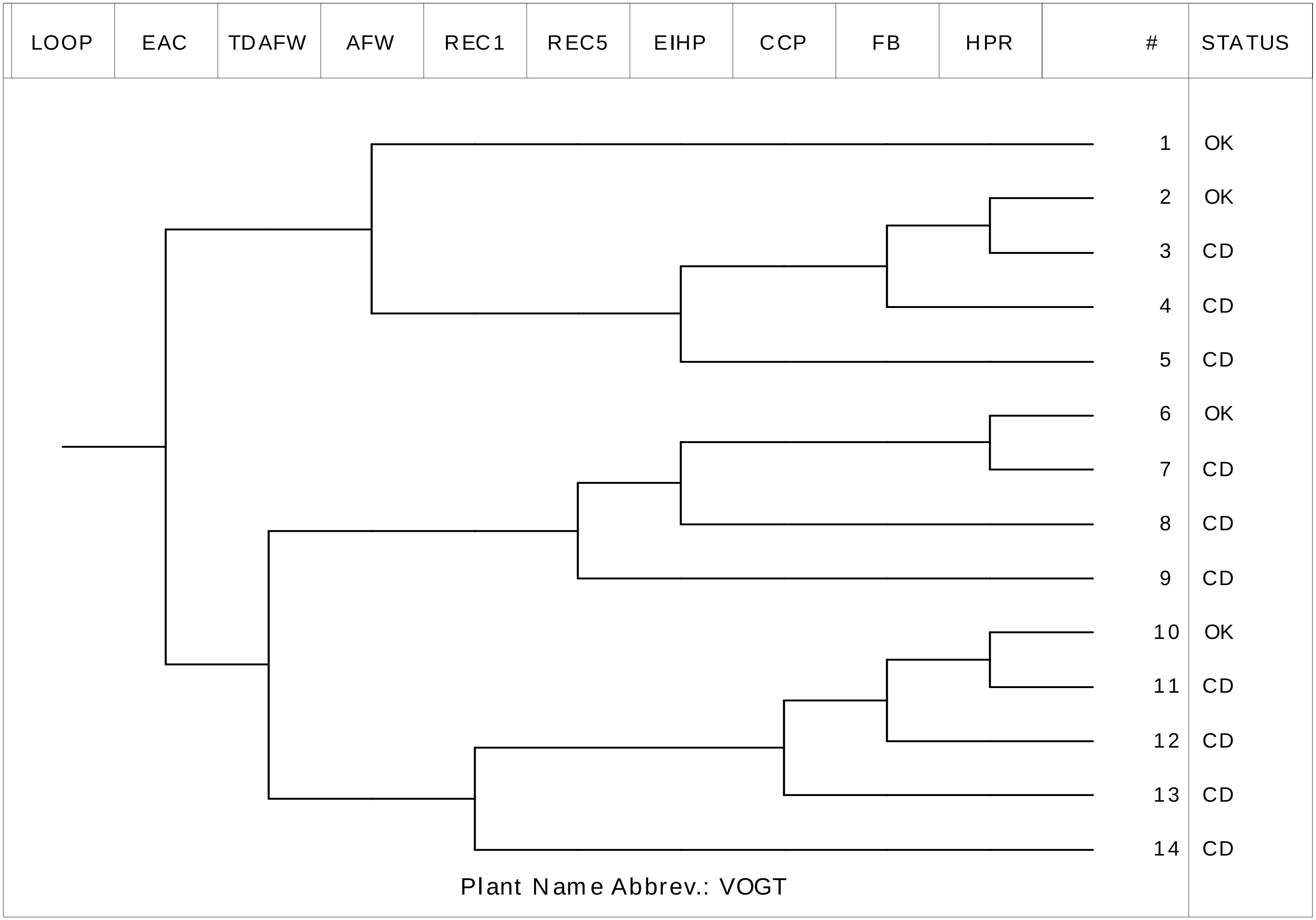
Plant Name Abbrev.:VOGT



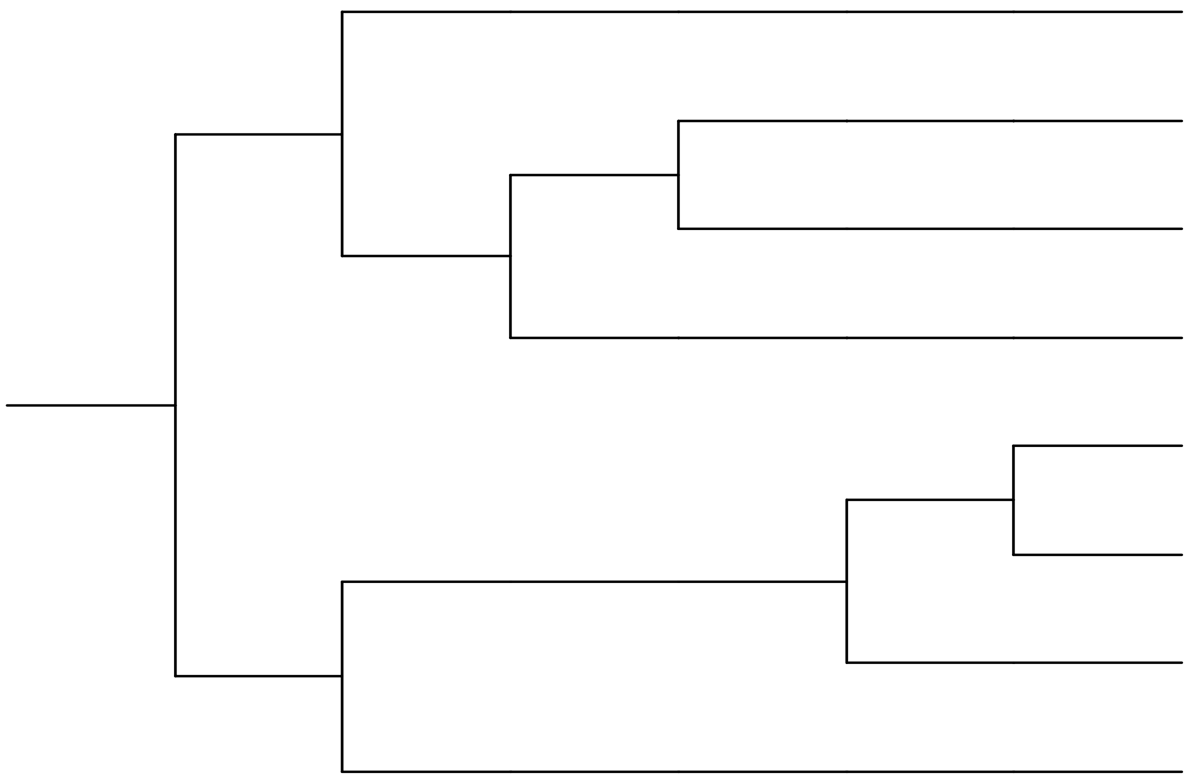
Plant Name Abbrev.: VOGT


MLOCA
EIHP
AFW
DEPR
ACC
LPI
HPR
LPR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 OK
7 CD
8 OK
9 CD
10 CD
11 CD
12 CD
13 CD
Plant Name Abbrev.: VOGT


LLOCA
ACC
LPI
LPR
#
STATUS
1
OK
2
CD
3
CD
4
CD
Plant Name Abbrev.: VOGT


LOOP
EAC
TDAFW
AFW
REC1
REC5
EIHP
CCP
FB
HPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 CD
9 CD
10 OK
11 CD
12 CD
13 CD
14 CD
Plant Name Abbrev.: VOGT

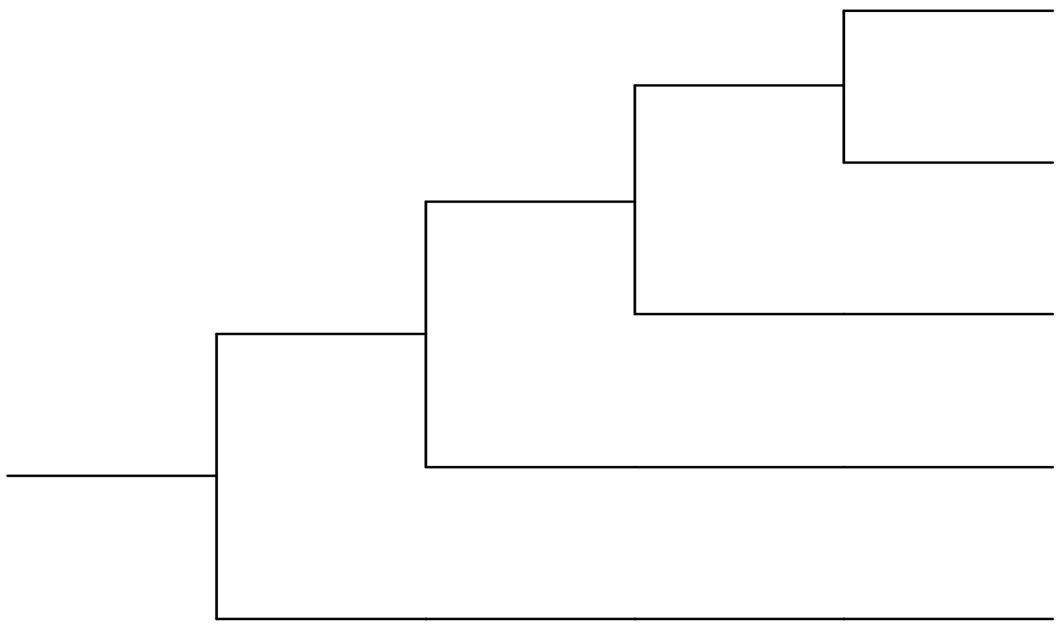
SGTR	AFW	EIHP	ISOL	EQ	FB	HPR	#	STATUS



#	STATUS
1	OK
2	OK
3	CD
4	CD
5	OK
6	CD
7	CD
8	CD

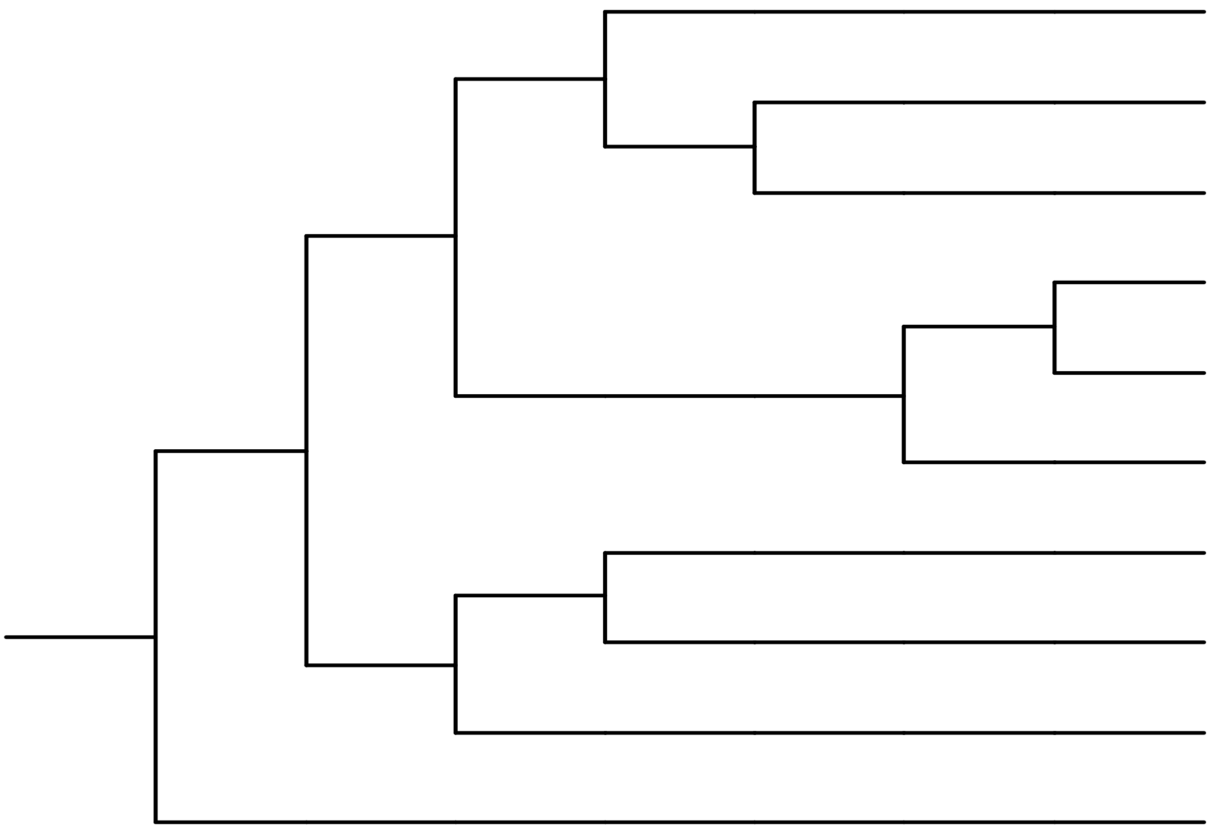
Plant Name Abbrev.: VOGT

ATWS	TTP	AFW	SRV	EMBO	#	STATUS
					1	OK
					2	CD
					3	CD
					4	CD
					5	CD

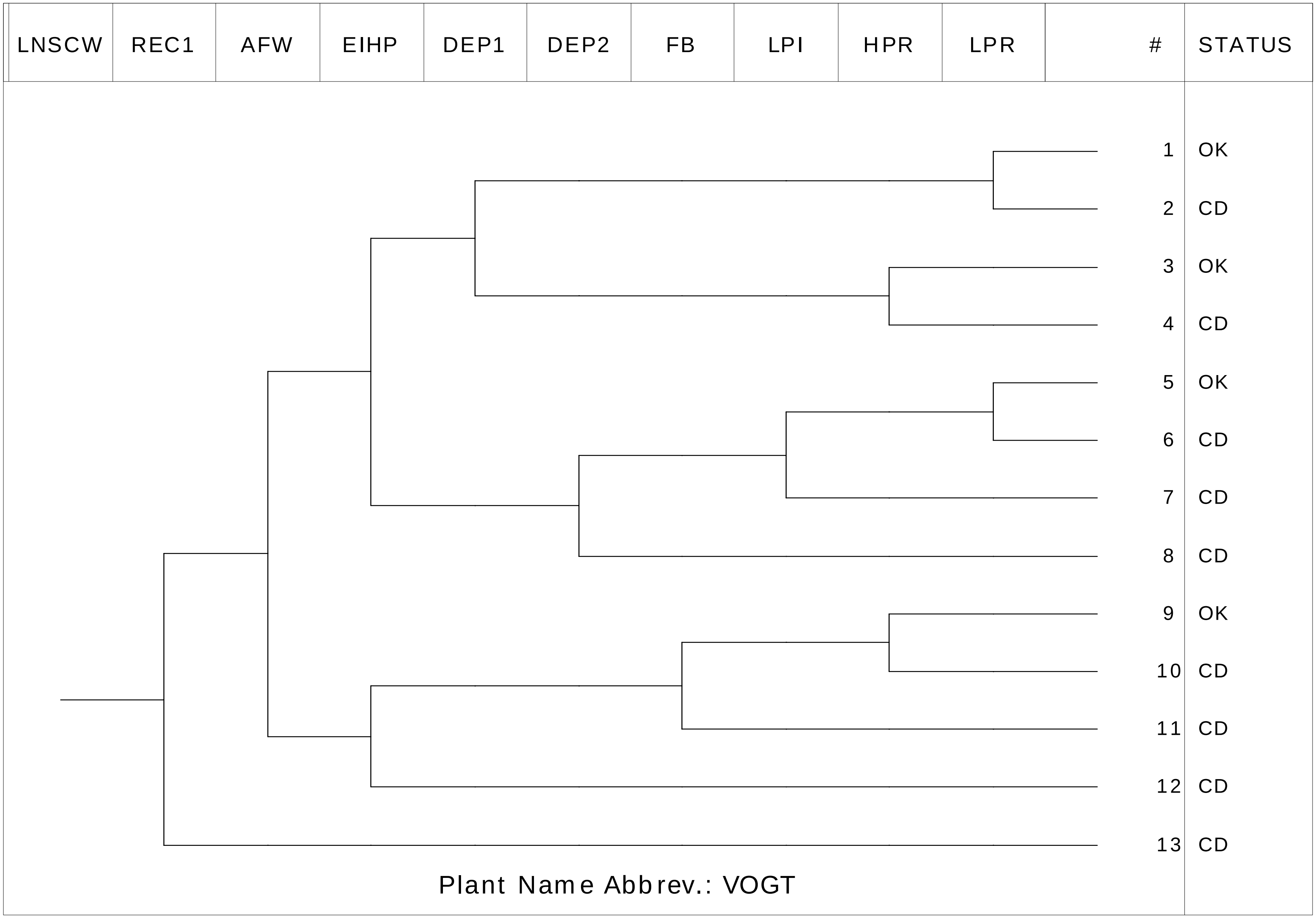


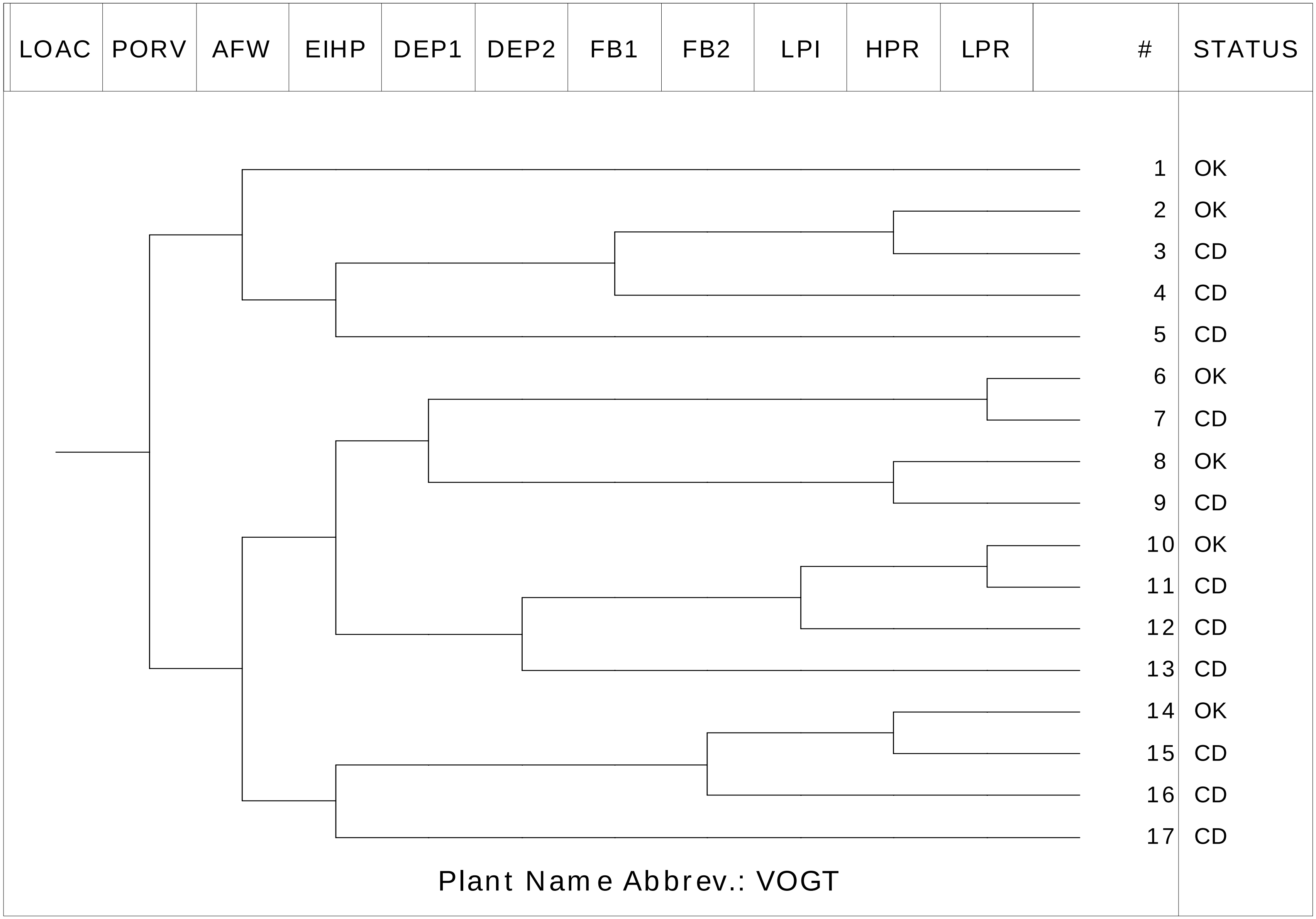
Plant Name Abbrev.: VOGT

MSLB	MSIV	EIHP	AFW	FWVC	STIN	FB	HPR	#	STATUS
								1	OK
								2	OK
								3	CD
								4	OK
								5	CD
								6	CD
								7	OK
								8	CD
								9	CD
								10	CD



Plant name abbrev.: VOGT


LNSCW
REC1
AFW
EIHP
DEP1
DEP2
FB
LPI
HPR
LPR
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 OK
6 CD
7 CD
8 CD
9 OK
10 CD
11 CD
12 CD
13 CD
Plant Name Abbrev.: VOGT


LOAC
PORV
AFW
EIHP
DEP1
DEP2
FB1
FB2
LPI
HPR
LPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 OK
9 CD
10 OK
11 CD
12 CD
13 CD
14 OK
15 CD
16 CD
17 CD
Plant Name Abbrev.: VOGT

2. RESOLUTION AND DISPOSITION OF COMMENTS

This section is composed of two subsections. Subsection 2.1 summarizes the generic assumptions that were used for developing the SDP worksheets for the PWR plants. These guidelines were based on the plant-specific comments provided by the licensee on the draft SDP worksheets and further examination of the applicability of those comments to similar plants. These assumptions which are used as guidelines for developing the SDP worksheets help the reader better understand the worksheets' scope and limitations. The generic guidelines and assumptions for PWRs are given here. Subsection 2.2 documents the plant-specific comments received on the draft version of the material included in this notebook and their resolution.

2.1 GENERIC GUIDELINES AND ASSUMPTIONS (PWRs)

The following generic guidelines and assumptions were used in developing the SDP worksheets for PWRs. These guidelines and assumptions were derived from a review of the licensee's comments, the resolutions of those comments, and the applicability to similar plants.

1. Assignment of plant-specific IEs into frequency rows:

Transient (Reactor trip) (TRANS), transients without PCS (TPCS), small, medium, and large LOCA (SLOCA, MLOCA, LLOCA), inadvertent or stuck-open PORV/SRV (SORV), main steam and feedwater line break (MSLB), anticipated transients without scram (ATWS), and interfacing system LOCAs (ISLOCA) are assigned into rows based on a consideration of the industry-average frequency. Plant-specific frequencies are considered for loss of offsite power (LOOP) and special initiators, and are assigned to the appropriate rows in Table 1.

2. Stuck open PORV/SRV as an IE in PWRs:

This event typically is not modeled in PRAs/IPEs as an initiating event. The failure of the PORVs/SRVs to re-close after opening is typically modeled within the transient event trees subsequent to the initiators. In addition, the intermittent failure or excessive leakage through PORVs as an initiator, albeit with much lower frequency, needed to be considered. To account for such failures and to keep the transient worksheets simple in the SDP, a separate worksheet for the SORV initiator was set up to explicitly model the contribution from such failures. This SDP worksheet, and the associated event tree, is similar to that of SLOCA. The frequency of PORV to re-close depends on the status of pressurizer. If the pressurizer is solid, then the frequency would be higher than the case in which the pressurizer level is maintained. Typically, this depends on early availability of secondary heat removal. However, the frequency for the SORV initiator is generically estimated for all PWR plants in Table 1.

3. Inclusion of special initiators:

The special initiators included in the worksheets are those applicable to this plant. A separate worksheet is included for each of them. The applicable special initiators are primarily based on the plant-specific IPEs/PRAs. In other words, the special initiators included are those modeled in the IPEs/PRAs unless shown to be negligible contributors. In some cases, a particular special initiator may be added for a plant even if it is not included in the IPE/PRA, if it is included in other plants of similar design, and is considered applicable for the plant. However, no attempt is made at this time to have a consistent set of special initiators across similarly designed plants. Except for the interfacing system LOCA (ISLOCA), if the occurrence of the special initiator results in a core damage, i.e., no mitigation capability exists for the initiating event, then a separate worksheet is not developed. For such cases, the inspection's focus is on the initiating event and the risk implication

of the finding can be directly assessed. For ISLOCA, a separate worksheet is included noting the pathways that can lead to it.

4. Inclusion of systems under the support system column of the Initiators and System Dependency Table:

This Table shows the support systems for the support- and frontline systems. The intent is to include only the support systems, and not the systems supporting that support system, i.e., those systems whose failure will result in failure of the system being supported. Partial dependency, e.g., a backup system, is not included. If they are, this should be so noted. Sometimes, some subsystems on which inspection findings may be noted were included as a support system, e.g., the EDG fuel oil transfer pump as a support system for EDGs.

5. Coverage of system/components and functions included in the SDP worksheets:

The Initiators and System Dependency Table includes systems and components which are included in the SDP worksheets and those which can affect the performance of these systems and components. One-to-one matching of the event tree headings/functions to that included in the Table was not considered necessary.

6. Crediting of non-safety related equipment:

SDP worksheets credit or include safety-related equipment and also, non-safety related equipment, as used, in defining the accident sequences leading to core damage. In defining the success criteria for the functions needed, the components included are those covered under the Technical Specifications (TS) and the Maintenance Rule (MR). Credits for other components may have been removed in the SDP worksheets.

7. No credit for certain plant-specific mitigation capability:

The significance determination process (SDP) screens inspection findings for Phase 3 evaluations. Some conservative assumptions are made which result in not crediting some plant-specific features. Such assumptions are usually based on comparisons with plants of similar design, and they help to maintain consistency across the SDP worksheets for similar plant designs.

8. Crediting system trains with high unavailability:

Some system component/trains may have unavailability higher than 1E-2, but they are treated similarly to other trains with lower unavailability in the range of 1E-2. In this screening, this approach is considered adequate to keep the process simple. An exception is made for steam-driven components which are designated as Automatic Steam Driven (ASD) train with a credit of 1E-1.

9. Treating passive components (of high reliability) the same as active components:

Passive components, namely accumulators, are credited similarly to active components, even though they exhibit higher reliability. Considering the potential for common-cause failures, the reliability of a passive system is not expected to differ by more than an order of magnitude from active systems. Pipe failures were excluded, except as part of initiating events where the appropriate frequency is used. Accordingly, a separate designation for passive components was not considered necessary.

10. Crediting accumulators:

SDP worksheets assume the loss of the accumulator unit associated with the failed leg in LOCA scenarios. Accordingly, in defining the mitigation capability for the accumulators, the worksheets refer to the remaining accumulators. For example, in a plant with 4 accumulators with a success criteria of 1 out of 4, for large LOCA the mitigation capability is defined as 1/3 remaining accumulators (1 multi-train system), assuming the loss of the accumulator in the failed leg. For a plant with a success criteria of 2 out of 4 accumulators, the mitigation capability is defined as 2/3 remaining accumulators (1 multi-train system).

The inspection findings are then assessed as follows (using the example of the plant with 4 accumulators and success criteria of 2 out of 4):

4 Acc. Available	Credit=3
3 Acc. Available (1 Acc. is considered unavailable, based on inspection findings)	Credit=2
< 3 Acc. Available (2 or more Acc. are considered unavailable, Based on inspection findings)	Credit=0

11. Crediting operator actions

The operator's actions modeled in the worksheets are categorized as follows: operator action=1 representing an error probability of 5E-2 to 0.5; operator action=2 representing an error probability of 5E-3 to 5E-2; operator action=3 representing an error probability of 5E-4 to 5E-3; and operator action=4 representing an error probability of 5E-5 to 5E-4. Actions with error probability > 0.5 are not credited. Thus, operator actions are associated with credits of 1, 2, 3, or 4. Since there is large variability in similar actions among different plants, a survey of the error probability across plants of similar design was used to categorize different operator actions. From this survey, similar actions across plants of similar design are assigned the same credit. If a plant uses a lower credit or recommends a lower credit for a particular action compared to our assessment of similar action based on plant survey, then the lower credit is assigned. An operator's action with a credit of 4, i.e., operator action=4, is noted at the bottom of the worksheet; the corresponding hardware failure, e.g., 1 multi-train system, is defined in the mitigating function.

12. Difference between plant-specific values and SDP designated credits for operator actions:

As noted, operator actions are assigned to a particular category based on a review of similar actions for plants with similar design. This results in some differences between plant-specific values and credit for the action in the worksheet. The plant-specific values are usually noted at the bottom of the worksheet.

13. Dependency among multiple operator actions:

IPEs or PRAs, in general, account for dependencies among the multiple operator actions that may be applicable. In the SDP screening approach, if multiple actions are involved in one function, then the credit for the function is designated as one operator action to the extent possible, considering the dependency involved.

14. Crediting the standby high-pressure pump:

The high-pressure injection system in some plants consists of three pumps with two of them auto-aligned and the third spare pump requiring manual action. The mitigating capability then is defined as : ½ HPI trains or use of a spare pump (1 multi-train system). Also, a footnote is added to reflect that the use of a spare pump could be given a credit of 1 (i.e.,1E-1) as a recovery action.

15. Emergency AC Power:

The full mitigating capability for emergency AC could include dedicated Emergency Diesel Generators (EDG), Swing EDG, SBO EDG, and finally, nearby fossil-power plants. The following guidelines are used in the SDP modeling of the Emergency AC power capability:

a) Describe the success criteria and the mitigation capability of dedicated EDGs.

b) Assign a mitigating capability of “operator action=1" for a swing EDG. The SDP worksheet assumes that the swing EDG is aligned to the other unit at the time of the LOOP (in a sense a dual unit LOOP is assumed). The operator, therefore, should trip, transfer, re-start, and load the swing EDG.

c) Assign a mitigating capability of “operator action=1" for an SBO EDG similar to the swing EDG. Note, some of the PWRs do not take credit for an SBO EDG for non-fire initiators. In these cases, credit is not given.

d) Do not credit the nearby power station as a backup to EDGs. The offsite power source from such a station could also be affected by the underlying cause for the LOOP. As an example, overhead cables connecting the station to the nuclear power plant also could have been damaged due to the bad weather which caused the LOOP. This level of detail should be left for a Phase 3 analysis.

16. Treatment of HPR and LPR:

The operation of both the HPR and LPR rely on the operation of the RHR pumps and the associated heat exchangers. Therefore, failure of LPR could imply failure of both HPR and LPR. A sequence which contains failure of both HPR and LPR as independent events will significantly underestimate the CDF contribution. To properly model this configuration within the SDP worksheets, the following procedure is used. Consider the successful depressurization and use of LPR as the preferred path. HPR is credited when depressurization has failed. In this manner, a sequence containing both HPR and LPR failures together is not generated.

17. SGTR event tree:

Event trees for SGTR vary from plant to plant depending on the size of primary-to-secondary leak, SG relief capacity, and the rate of rapid depressurization. However, there are several common functional steps that are addressed in the SDP worksheet: early isolation of the affected SG, initiation of primary cool-down and depressurization, and prevention of the SG overfill. These actions also include failure to maintain the secondary pressure below that of Main Steam safety valves which could occur either due to the failure of the relief valves to open or the operator's failure to follow the procedure. Failure to perform this task (sometimes referred to as early isolation and equalization) is assumed to cause continuous leakage of primary outside the containment. The success of this step implies the need for high-pressure makeup for a short period, followed by depressurization and cooldown for RHR entry (note, relief valves are assumed to re-close when primary pressure falls below that of the secondary). If the early makeup is not available or the operator fails to perform early isolation and equalization, rapid depressurization to RHR entry is usually assumed. This would typically require some kind of intermediate- or low-pressure makeup. Finally, depending on the size of the Refueling Water Storage Tank (RWST), sometimes it would be necessary to establish makeup to the RWST to allow sufficient time to enter the RHR mode.

18. ATWS scenarios:

The ATWS SDP worksheet assumes that these scenarios are not recoverable by operator actions, such as a manual trip. The failure of the scram system, therefore, is not recoverable, neither by the actuation of a back-up system nor through the actuation of manual scram. The initiator frequency, therefore, should only account for non-recoverable scrams, such as mechanical failure of the scram rods.

19. Recovery of losses of offsite power:

Recovery of losses of offsite power is assigned an operator-action category even though it is usually dominated by a recovery of offsite AC, independent of plant activities. Furthermore, the probability of recovery of offsite power in "X" hours (for example 4 hours) given it is not recovered earlier (for example, in the 1st hour) would be different from recovery in 4 hours with no condition. The SDP worksheet uses a simplified approach for treating recovery of AC by denoting it as an operator

action=1 or 2 depending upon the HEP used in the IPE/PRA. A footnote highlighting the actual value used in the IPE/PRA is provided, when available.

20. RCP seal LOCA in a SBO:
The RCP seal LOCA in a SBO scenario is included in the LOOP worksheet. RCP seal LOCA resulting from loss of support functions is considered only if the loss of support function is a special initiator. The dependencies of RCP seal cooling are identified in Table 2.

21. RCP Seal LOCA for Westinghouse Plants during SBO Scenarios:

The modeling of the RCP seal failures upon loss of cooling and injection as occurs during SBO scenarios has been the subject of many studies (e.g., BNL Technical report W6211-08/99 and NUREG/CR-4906P). These studies are quite complex and assign probabilities of seal failure as a function of time (duration of SBO) and the associated leak rates. The leak rates, in turn, will determine what would be the safe period for recovery of the AC source and the use of SI pumps before core uncovery and damage. On the contrary, the SDP worksheets simplify the analysis of the RCP seal LOCA during the SBO scenarios using the following two assumptions: (1) The probability of catastrophic RCP seal failure is assumed to be 1 if the SBO lasts beyond two hours, and (2) Given a catastrophic seal LOCA, the available time prior to core damage for recovery of offsite power and establishing injection is about two hours. Therefore, in almost all cases, to prevent a core damage, a source of AC should be recovered within 4 hours in SBO scenarios.

22. Tripping the RCP on loss of CCW:

Upon loss of CCW, the motor cooling will be lost. The operation of RCPs without motor cooling could result in overheating and failure of bearings. Bearing failure, in turn, could cause the shaft to vibrate and thereby result in the potential for seal failure if the RCP is not tripped. In Westinghouse plants, the operator is instructed to trip the RCPs early in the scenario (from 2 to 10 minutes after detecting the loss of cooling). Failure to perform this action is conservatively assumed to result in seal failure and, potentially in a LOCA. This failure mechanism (occurrence of seal LOCA) due to failure to trip the RCPs upon loss of cooling is not considered likely in some plants, whereas it has been modeled explicitly in other plants. To ensure consistency, the trip of the RCP pumps are modeled in the SDP worksheets, and the operator failure to do this is assumed to result in a LOCA. In many cases, the failure to trip RCP following a loss of CCW results in core damage.

23. Hot leg/Cold leg switchover:

The hot leg to cold leg switchover during ECCS recirculation is typically done to avoid boron precipitation. This is typically part of the procedure for PWRs during medium and large LOCA scenarios. Some IPEs/PRAs do not consider the failure of this action as relevant to core damage. For plants needing the hot /cold switchover, it usually can only be accomplished with SI pumps and the ECCS recirculation also uses the SI pumps.

2.2 RESOLUTION OF PLANT-SPECIFIC COMMENTS

The Licensee provided detailed and well-organized comments on the draft worksheets. These comments were very useful to update this document. The special initiators, plant response, and the initiator impacts were discussed. Other items included as a part of NRC review process were also discussed. Several questions were raised and written responses were received from the licensee subsequent to the meeting. The licensee responses were reviewed and incorporated into the SDP worksheet to the extent possible within the framework, scope, and limitations of the SDP worksheets. The licensee's comment and feed back have significantly contributed to the improvement of this document.

(1) Licensee's comments on Table 2, "Initiators and System Dependency for Alvin W. Vogtle, Units 1 & 2" reflecting the up to date plant-specific system interactions, clarification notes, and plant-specific acronyms were incorporated.

(2) Licensee's comments on the SDP worksheets were incorporated.

(3) Hot leg recirculation was removed from the medium and large LOCAs. The licensee stated that hot leg recirculation is not needed to prevent core damage.

(4) The licensee removed the termination of SI in an MSLB, but it did not discuss the basis for such removal. For this reason, we kept the termination of SI in an MSLB.

(5) The licensee provided useful comments on the event trees included in the previous version of the SDP document, but it did not provide information on the event trees related to failure of support systems. The items below discuss the impact of the loss of some support systems, based on information contained in the IPE.

    (1) This discussion on loss of ACCW is based on the information given on the IPE's "Initiator to System Dependency Matrix" (page C-3), and on note 27 of page C-6. The ACCW provides cooling for the following components: RCP thermal barrier heat exchanger and the RCP bearing oil coolers, the CVCS' letdown heat exchanger, the CVCS' excess letdown heat exchanger, the CVCS' seal water heat exchanger, and the CVCS' positive displacement charging pump and motor cooler. Loss of cooling to the RCP bearing oil coolers could result in overheating and failure of bearing. Bearing failure, in turn, could cause the shaft to vibrate and thereby result in the potential for seal failure if the RCP is not tripped.

    Cooling to the RCP thermal barrier heat exchanger is lost, but RCP seal cooling would be provided by the centrifugal charging pumps. Hence, we assume that an RCP seal LOCA would not occur as long as the CCPs provide RCP seal cooling. We also assume that the loss of cooling to the CVCS' seal water heat exchanger does not have a significant impact

on the seal cooling provided by the CCPs because they take suction from the volume control tank (VCT). The frequency of loss of ACCW is 1.3E-3 / year (IPE, page 3-6).

According to this discussion, there are two possibilities for an RCP seal LOCA to occur subsequent to a loss of ACCW: 1) if the operators fail to trip the RCPs (operator action = 3), and 2) if 1 / 2 centrifugal charging pumps (1 multi-train system) fail to provide RCP seal cooling. Hence, the frequency of a loss of ACCW-induced RCP seal LOCA is of the order of 10^{-6}, which is less than the frequency of a small LOCA or a medium LOCA. Accordingly, we did not model a loss of ACCW because it is bounded by small LOCA and medium LOCA. A sequence with the initiating event loss of ACCW was not found in the 100 dominant accident sequences presented in the IPE (Table 3.4-3). Loss of ACCW contributes 0.03% to the total CDF.

(2) This discussion on loss of IA is based on the information given on the IPE's "Initiator to System Dependency Matrix" (page C-3), and on note 23 of page C-6. The impact on the plant of a loss of IA is the following: the feedwater regulating and bypass valves fail closed, the MSIVs eventually close, the steam dumps fail closed, letdown is isolated, and the normal charging flow control valves fail open. The frequency of loss of IA is 2.4E-2 / year (IPE, page 3-6). Hence, loss of Instrument Air is similar to and bounded by Transients with Loss of PCS (TPCS). Loss of IA contributes 0.28% to the total CDF.

(3) This discussion on loss of one train of Control Building (CB) ESF Electrical HVAC is based on the information given on the IPE's "Initiator to System Dependency Matrix" (page C-3), and on note 28 of page C-6. The impact on the plant of a loss of one train of Control Building (CB) ESF Electrical HVAC is the loss of room cooling to the corresponding train of Class 1E electrical power (4160 V AC buses, 480 V AC buses and MCCs, 125 V DC buses, and 120 V AC panels). This would have a delayed impact on failure of these systems. The frequency of loss of one train of Control Building (CB) ESF Electrical HVAC is 4.1E-3 / year (IPE, page 3-6). Hence, we assume that this loss is bounded by the loss of one 125 V DC bus (LBDC). Loss of one train of Control Building (CB) ESF Electrical HVAC contributes 0.01% to the total CDF.

(6) Loss of CCW is not modeled by IPE as an initiating event. According to IPE's Figure C-3, "Support System Dependency Matrix" (page C-11), CCW only supports the RHR pumps.

(7) There are two Class 1E ESF 4.16 kV buses per unit. Loss of one ESF 4.16 kV bus is not modeled by IPE as an initiating event. Each bus supports one train of safeguards equipment (IPE, page C-12, notes 1 and 2), so loss of one ESF 4.16 kV bus would cause the loss of one train of safeguards equipment. Hence, it appears that a loss of one ESF 4.16 kV bus is bounded by the loss of one 125 V DC bus (LBDC).

(8) The remaining comments from the licensee are addressed mainly in Sections 1, "Information Supporting Significance Determination Process (SDP)", and 2.1, "Generic Guidelines and Assumptions (PWRs)" of this document.

REFERENCES

1. NRC SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007), March 22, 1999.

2. Georgia Power Company, "Alvin W. Vogtle Electric Generating Plant, Units 1 and 2 — Individual Plant Examination Report," December 1992.

(3) Note from S. M. Wong to J. G. Ibarra, “Vogtle SDP Worksheets”, contains Southern Nuclear Company’s comments on the Vogtle SDP Worksheets, May 10, 2000.